Entergy

Entergy Nuclear Northeast
Indian Point Energy Center
295 Broadway, Suite 1
P.O. Box 249
Buchanan, NY 10511-0249
Tel 914 734 5340
Fax 914 734 5718

Fred Dacimo
Vice President, Operations

May 27, 2003

Re: Indian Point Unit Nos. 1 and 2
Docket Nos. 50-003 and 50-247
NL-03-084

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station O-P1-17
Washington, DC 20555-0001

SUBJECT: 10 CFR §50.59 (d) Report for Indian Point Unit Nos. 1 and 2

Pursuant to 10 CFR §50.59 (d)(2), enclosed please find a summary report of the changes, tests and experiments implemented at Indian Point Unit Nos. 1 and 2 between January 3, 2001 and November 27, 2002, or utilized in support of the UFSAR update. The summaries of Safety Evaluations (SEs) and 50.59 Evaluations (RE) set forth in the report represent the changes in the facilities, changes in procedures and tests and experiments implemented pursuant to 10 CFR §50.59. Attachment 1 includes a listing of the above mentioned evaluations. Attachment 2 provides a summary of those evaluations Implemented in the period defined above.

Entergy is making no new commitments in this submittal.

Sincerely,

Fred Dacimo
Vice President, Operations
Indian Point Energy Center

IE47

Attachments

cc: Hubert J. Miller
Regional Administrator - Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Patrick D. Milano, Project Manager
Project Directorate I
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop 0-8-C2
Washington, DC 20555-0001

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Indian Point Unit 2
P.O. Box 38
Buchanan, NY 10511-0308

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Indian Point Unit 3
P.O. Box 337
Buchanan, NY 10511-0308

ATTACHMENT 1 TO NL-03-084

50.59 REPORT LISTING

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 1 and 2
DOCKET NO. 50-003 and 50-247

50.59 Evaluation Number	Rev. No.	50.59 EVALUATION TITLE
90-120-MD	0	Replace Condenser No. 21
91-140-MD	0	IP2 – SI Pump Suction – Over Pressure Protection
92-125-MD	0	Replace Condenser #22 and Feedwater Heaters #21B and 22B
94-118-MD	1	IP #2 Rehabilitation of Cathodic Protection of Dock Sheet Piles – Phase II
94-168-DE	0	Replacement of Pratt Butterfly Valves
96-229-MD	0	Addition of Bypass Switches for Nuclear Instrumentation System Dropped Rod Turbine Runbacks
96-241-GM	1	Generic Replacement of Horizontal Centrifugal Pump and Motor Assemblies Up to 25 HP for Liquid Applications
97-011-MD	0	City Water Backflow Preventers
98-021-MM	0	Fire Water Storage Tank Heaters
98-174-MM	0	Roof Replacement to the Auxiliary Boiler Feed Water Building
98-281-MM	0	Fire Water Storage Tank Level Instrumentation Modification
98-340-MM	2	Relocation of Service Water Pressure Gauges for Emergency Diesel Generator Coolers
98-384-MD	0	Replacement of SW Strainer Blowdown Valves SWN-617 Through SWN-622

50.59 Evaluation Number	Rev. No.	50.59 EVALUATION TITLE
99-031-MM	1	Installation of a 3-Hour Rated Fire Door in Fire Zone 6A (El. 80', PAB) and Removal of Radiation Monitor R-8
99-112-MM	1	Remove Reactor Coolant System Sample Booster Pump
99-222-MD	0	Removal of Regenerative Tank 21, Regenerative Pump 21 and Associated Piping and Electrical Appurtenances
99-291-MM	0	NYPA/Con Ed Joint Effort on Water Treatment Plant
99-379-MM	0	Charging Pumps Oil Level Sightglass Installation
00-123-MD	1	Replacement of Condenser Neck Feedwater Heater Tube Bundles
00-369-MM	0	Drain Traps Replacement of Steam Jet Air Ejector Condensers for Secondary Side pH Control
00-598-MD	0	Reactor Coolant System Redundant Level Measuring System at Draindown
00-611-MD	0	Hotwell (Sextant) On-Line Sampling Relocation
00-613-MD	0	Relocation Feedwater Sampling and Cooling Upgrade
00-736-MM	0	Hotwell Sextant and Steam Generator Blowdown Alarm Conversion to Sodium
00-764-MM	0	Chlorination System Upgrade
01-177-TM	0	Temporary Facility Change for Repair of the City Water Header
01-225-EV	0	Incorporation of Revised SQUG Methodology
01-441-PR-00-RE	0	Steam Generator Moisture Carryover Test
02-245-EV-00-RE	0	Use of Unit 3 Appendix R Diesel Generator to Satisfy Unit 2 Technical Specification 3.7.C.3
02-412-CL-00-RE	0	Containment Peak Pressure for a Postulated Steam Line Break

50.59 Evaluation Number	Rev. No.	50.59 EVALUATION TITLE
02-420-CL-00-RE	0	Evaluation for a Potential Recritiality
02-498-EV-00-RE	0	Remote Examination and Removal of Foreign Objects from Steam Generator

ATTACHMENT 2 TO NL-03-084

50.59 REPORT
SUMMARY LISTING

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 1 and 2
DOCKET NO. 50-003 and 50-247

50.59 Summary of Changes, Tests, and Experiments

50.59/Design Change Evaluation Number	Rev. No	TITLE

90-120-MD	0	Replace Condenser No. 21

This Modification replaced the internals of Condenser #21 with titanium tubing yielding a more reliable and leak resistant system. No New failure modes were introduced and any failure or malfunction of the modified system would be identical to that in the existing system. In addition, this modification has the beneficial effect of enhancing secondary water chemistry. This Modification was part of the overall plan to reduce copper and oxygen concentrations in the water to minimize or preclude cracking in steam generator girth welds and feedwater nozzles. The safety-related function of the air ejector and its radiation monitor (to detect steam generator tube leaks and, as necessary, divert effluent gases from the plant vent stack to containment) remains unchanged,

91-140-MD	0	IP2 – SI Pump Suction – Over Pressure Protection

Check valve leakage during operation of 21 and/or 23 Safety Injection (SI) pumps operating with Motor Operated Valves (MOV) 887 A / B closed could cause a pressure increase above the design pressure for 22 SI pump suction piping. This Modification installed a ¾" 304 Stainless relief line from valves S-11 to S-6 to prevent over pressurization of SI pump 22 suction piping. Additionally, a new isolation valve #7352 was added in the relief line to allow for isolation in the event of a break of the normal recirculation line during recirculation. Valves S-6, S-11 and 7352 are sealed open during normal operation.

92-125-MD	0	Replace Condenser #22 and Feedwater Heaters #21B and 22B

This Modification installed a modular titanium condenser, new waterboxes in 22# condenser along with new stainless steel feedwater heater internals in heaters 21# and 22B.

94-118-MD	1	IP #2 Rehabilitation of Cathodic Protection of Dock Sheet Piles – Phase II

This Modification replaced the original cathodic protection system provided at the dock with a new system for the dock sheet piling that is larger in size with two Rectifiers (60Vdc at 180 amps and 50 Vdc at 80 amps) and eight anodes. The original protection consisted of five anode ground beds placed at the river bottom on each side of the intake structure and a single 50 Vdc 64 amps. The old anodes were either consumed or lost.

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50.59/Design Change Evaluation Number	Rev. No	TITLE

96-229-MD	1	Addition of Bypass Switches for Nuclear Instrumentation System Dropped Rod Turbine Runbacks

Bypass switches have been installed so that a turbine runback on rod drop no longer occurs automatically (Modification FIX-93-09927-M, SE 96-229-MD). The subject Modification installed bypass switches which can defeat the turbine runback on a Nuclear Instrumentation System (NIS) dropped rod signal. Plant Check-Off List 13.1 requires the switches to be placed in the defeat position so this turbine runback signal is normally bypassed. Bypassing the runback signal improves plant reliability and prevents unexpected reductions in turbine generator load. The accident analysis indicates that the consequences of a dropped rod accident even without the turbine runback are within the criteria for this accident. Additionally, no other accidents rely on the NIS runback.

96-241-GM	1	Generic Replacement of Horizontal Centrifugal Pump and Motor Assemblies Up to 25 HP for Liquid Applications

This Modification allows for the replacement of pump and motor assemblies, Class A and Non-Class A, provided that the installation meets the conditions required. Minor changes to piping, piping supports, foundations, conduits, electrical protection and conduit supports are permitted. Requirements and specifications for the replacement assemblies will be the same as for the original assemblies.

97-011-MD	0	City Water Backflow Preventers

This Modification installed three back flow preventers in the three eight inch lines supplying City Water to the 1,500,000 gallon Plant City Water Storage Tank. The purpose of these back flow preventers is to prevent contamination of the potable water system and subsequently the public water system.

98-021-MM	0	Fire Water Storage Tank Heaters

This Modification replaced the non-functioning freeze protection heater system for the Fire Water Storage Tank with a double output / dual setting controller. Each controller output will actuate one 150 Kw heater bank. Additionally, it adds set points for the breakers in the heater control Cubicle and the substation A Breaker. The Modification also replaces a leaking thermowell. This Modification does NOT change any safety function and it does increase reliability by the elimination of a heat source (SCR) from the control panel.

50.59 Summary of
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50.59/Design Change Evaluation Number	Rev. No	TITLE
98-174-MM	0	Roof Replacement to the Auxiliary Boiler Feed Water Building

This modification replaced the roofing membrane system of the Auxiliary Boiler Feed Water Building. Roof leaks and inspection justified the roof replacement. The roof was made water tight by application of a new single-ply membrane system that will protect the building interior from the elements. The new system is a fully adhered E.D.P.M. membrane system including roofing membrane, insulation, flashing, fascias, expansion joint, pitch pockets and walkways.

98-281-MM	0	Fire Water Storage Tank Level Instrumentation Modification

This Modification installed a new High Accuracy (Plus or Minus 1/8 inch) level transmitter that significantly reduced the instrument uncertainties for the level loop from 10.99 to 1.6 inches so that existing alarm settings do not overlap. The new sensor (LE-700) and transmitter (LT-700) are mounted inside the Diesel Fire Pump Building. The new transmitter provides four contacts for alarms and valve control signals. The transmitter is also equipped with a display panel that is used for programming during system setup and readout of the Fire Water Storage tank level. This system can also function as a backup to the Central Control Room (CCR) level indication. The Modification replaced the old analog level indicator with a new digital display with improved accuracy and readability.

98-340-MM	0	Relocation of Service Water Pressure Gauges for Emergency Diesel Generator Coolers

This Modification removed the existing pressure gauges PI-5657, PI-5677 and PI-5678 from Emergency Diesel Generator (EDG) #21, #22 and #23 indicating gauge panels and replaced them with more accurate gauges that are located closer to the coolers. The new pressure gauges were connected to the existing pressure tap connections, but with shorter tubing runs. The process range of the new gauges was reduced from 0-200 psig, (for EDG #21 and #23) and 0-160 psig (for EDG #22) to 0-100 psig to increase the accuracy of the Service Water system pressure reading. The new gauges are seismically qualified and installed near the EDG coolers.
This Modification also eliminated Temporary Facility Change (TFC) 98-013 and eliminated the previously identified discrepancy of 10 psig difference between the TFC and the indicating panel gauge.

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98-384-MD	0	Replacement of SW Strainer Blowdown Valves SWN-617 Through SWN-622

This Modification removed the motor operators on Motor Operated Valves (MOV) SWN-617 through SWN-622 and replaced them with manual throttling valves. These valves are installed on the blowdown lines of the Service Water Pump strainers. The associated cables for the MOVs were removed or retired in place. SWN-617 through SWN-622 have a history of failures and are obsolete.
The reduction in SW blowdown flow is required to maintain the required SW flow of the Essential Header during Design Basis Accident conditions. This requirement is met by throttling and locking the Strainer Outlet Valves to assure the valve position will not change in a seismic event. Safety Evaluation 98-322-EV documented acceptance of this position. The throttled positions are controlled and monitored periodically. The SW backwash piping has been tested in accordance with established plant procedures. The Technical Specification surveillance requirements and existing Technical Specification required surveillance related tests, calibrations or inspection procedures are not affected.

99-031-MM	1	Installation of a 3-Hour Rated Fire Door in Fire Zone 6A (El. 80', PAB) and Removal of Radiation Monitor R-8

This Modification replaced the existing motor operated door with a 3 hour rated fire door and removed the radiation monitor R-8. The area Radiation Monitor and shielding function of the door are no longer required because radioactive waste is no longer stored in this room. The room is now used for storage of both radioactive and non-radioactive equipment, material or components. A portable Radiation Monitor is used when required.

99-112-MM	1	Remove Reactor Coolant System Sample Booster Pump

This revision to Modification 99-112 provided the provisions for cutting and capping the sample lines associated with the retired Sample Booster Pump, in other than Cold Shutdown. Isolation can be established by isolating the RCS Sample Heat Exchangers. The installation of caps while on-line did not change or affect the plant system nor create an unsafe condition as the isolation valves were designed to provide positive isolation. The above changes did not create any significant change of scope.

99-222-MD	0	Removal of Regenerative Tank 21, Regenerative Pump 21 and Associated Piping and Electrical Appurtenances

This Modification retired and removed the waste disposal Regenerative Tank 21, Regenerative Pump 21, associated piping and electrical equipment under Modification FMX-99-12267-M as shown on drawing 9321-F-2719.

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99-291-MM	0	NYPA/Con Ed Joint Effort on Water Treatment Plant

This Modification provides Treated water to Indian Point Unit 2 (IP2) by providing to the Indian Point Unit 1 (IP1) Condensate Storage Tank (CST) an adequate inventory of high quality water for various uses and an emergency backup to Indian Point Unit 3 (IP3). The 6 inch Condensate line that has been changed was previously used to provide water from the IP1 CST to IP 3 for use. The new 6 inch Condensate line has been isolated from IP 3 and when required for Emergency use may be reopened to PROVIDE IP 3 with a backup supply. The safety related function of the IP2 CST has remained unaffected by this Modification.

99-379-MM	0	Charging Pumps Oil Level Sightglass Installation

This Modification installed new sight glass level indicators on each Charging Pump (21CHP, 22 CHP and 23CHP) oil crankcase drain line. This reduced the man-rem exposure by the operators while reading the Charging Pump oil Level. The new oil level indicators are enclosed on three sides by steel with two sides extended slightly beyond the tube on the open front to prevent breakage. The new sight glasses do not require the operators to remove a screw cap located on the oil refill line to check level in the crankcase. This Modification reduces the dose to the operators during inspections.

00-123-MD	1	Replacement of Condenser Neck Feedwater Heater Tube Bundles

This Modification replaced the 21 and 22 A, B and C Feedwater Heater tube bundles, the Gland Steam Condenser and the operating vent piping for the 21 and 22 A, B and C Feedwater Heat exchangers, during the 2000 Refueling Outage. Revision 1 of this modification changed the Normal Water Level and the low level alarms from 2.25 to 1.5 inches below Normal Water Level. The Feedwater Heaters and controls in this case are Non-Class A and located in the Turbine Building with no Class A SSC interface or proximity. The Setpoint changes are for thermal performance enhancement.

00-369-MM	0	Drain Traps Replacement of Steam Jet Air Ejector Condensers for Secondary Side pH Control

This Modification replaced the Drain Traps for the Steam Jet Air Ejector (SJAE) after-condenser drain lines, and re-established the flow path from the drain traps to the Main Condenser. These lines had been previously cut and capped due to secondary water chemistry considerations. With the replacement of the feedwater heaters the copper containing components have been removed from the secondary system, and higher pH

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levels are desired. This Modification facilitates maintaining higher pH levels by recirculating after-condenser drainage to the condenser.

00-598-MD	0	Reactor Coolant System Redundant Level Measuring System at Draindown

This Modification installed a new upper level tap connection to the pressurizer for the RCS Level Measuring System at Draindown through the existing isolation valve 537. The new upper level tap is now connected to the condensing tee for Pressurizer level transmitter, LT-459. The previous tap used for the RCS level transmitter was from the four inch vent line to the Power Operated Relief Valves (PORV), and was potentially susceptible to dynamic velocity effects during the creation of an RCS vacuum through the PORVs during RCS Vacuum fill and draindown. When the RCS Level Measuring system is not in use, valves 537B and 537C will be closed and the tubing end capped to eliminate any effect on RCS pressure boundry or pressurizer level transmitter, LT-459 during normal plant operation.

00-611-MD	0	Hotwell (Sextant) On-Line Sampling Relocation

This Modification provides state of the art on-line analytical analysis of samples from the condenser hotwells. The On-Line Monitoring Panel is used continuously during power operations. The analyzers and sampling components satisfy ASME and ASTM sampling requirements for pressure, temperature and flow (ASTM standards include but are not limited to the latest revisions of D1192 and D3370). Cooling equipment to reduce sample temperatures has also been installed. Justification for the Modification is that cation conductivity and sodium ion monitors in condensate are utilized as indications of cooling water leakage (EPRI NP-7382 Design and Operating Guidelines for Nuclear Power Plant Condensers).

00-613-MD	0	Relocation Feedwater Sampling and Cooling Upgrade

This Modification improved the on-line chemical analysis of samples from the high-pressure main feedwater system (outlet of 26 feedwater heaters). The system is to be used continuously during power operations. The analyzers and sampling components satisfy ASME and ASTM sampling requirements for pressure, temperature, and flow.
The analytical instrumentation for high-pressure main feedwater sampling was upgraded to Orion Oxygen Scavenger analyzers for hydrazine and ABB Dissolved Oxygen Monitors. Specific and cation conductivity measurement was upgraded to a Honeywell system. PH was upgraded to a Honeywell. The existing High Pressure Feedwater corrosion product panel was relocated to the new sample panel. This Modification also upgraded the cooling systems for the Heater Drain Tank and condensate corrosion produce sample line.

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00-736-MM	0	Hotwell Sextant and Steam Generator Blowdown Alarm Conversion to Sodium

This Modification converted alarms 1FAF window 1-5 and FCF window 1-11 from chloride to sodium alarm function. The stream generator blowdown chloride and sodium recorder wiring was exchanged to allow the sodium current loop to supply the CCR bistables that actuate the Blowdown Sodium Alarm. Additionally, the wiring for the Sextant sodium loop was swapped to supply the CCR bistables for the Sextant Sodium. This Modification also changed the alarm setpoints to align with the EPRI Guidelines.

00-764-MM	0	Chlorination System Upgrade

This Modification is designed to improve the efficiency and reliability of the Chlorination System. It consists replacing the existing equipment between the active Sodium Hypochlorite tank and the diffusers for Unit 2 Circulating Water, Unit 2 Service Water and Unit 1 River Water. Additionally, minor changes were made to the City Water system serving the Screenwell House to support Safety Shower and Eyewash station relocation.

Two new 100 percent capacity (0-500 gph) positive displacement pumps replaced the existing pumps to provide sodium hypochlorite solution to the Unit 2 Circulating Water Diffusers. Two new 100 percent low flow (0-57 gph) positive displacement pumps provide sodium hypochlorite solution to the Unit 1 River Water and Unit 2 Service Water diffusers. A third low capacity standby spare is provided for use for either River Water or Service Water. A 500 gallon day tank is provided to supply the pump skid with sodium hypochlorite solution. A centrifugal transfer pump is used to fill the day tank from the storage tank. A valve CL-1282 is installed in the suction line for the transfer pump to limit the potential for downstream leakage and isolate the pump from the static head of the storage tank when the pump is not operating. An interlock closes CL-1282 when the transfer pump is shutdown. Over pressure protection is supplied by relief valves set to lift well below the pressure rating of the piping.

All system components are classified as non-safety related and non-seismic. MCC-21 is the power supply for the system loads. There are no increases in loading of MCC-21 based on the limiting number of pumps that can be loaded simultaneously per System Operating Procedure SOP 22.4.

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01-177-TM	1	Temporary Facility Change for Repair of the City Water Header

This Temporary Facility Change (TFC) maintained an alternate supply of City Water from the City Water Storage Tank to Indian Point 2 while maintenance repairs of the existing City Water header were accomplished. The TFC routed water from the City Water Storage Tank to the Unit 1 City Water header. One limitation on the TFC was that Unit 3 was shutdown or that an alternate source of water to the Unit 3 Auxiliary Feedwater Pumps be in place.
Backup supplies from Fire Protection Water were maintained to provide cooling for the Charging and RHR pumps, which are normally provided with backup from City Water during the TFC. The backup supply did not impact the operability of the Fire Protection System since they did not exceed the maximum capacity of the system. No new failure modes were introduced by this TFC.

01-225-EV	0	Incorporation of Revised Seismic Qualification Users Group (SQUG) Methodology

Incorporated the SQUG methodology into the plant's licensing basis as an alternate means of performing seismic evaluation of SSCs. This evaluation provided the basis for adding the SQUG technique to the UFSAR.

01-441-PR-00-RE	0	Steam Generator Moisture Carryover Test

Activity Description:
This new procedure governs conducting a moisture carryover test to determine the amount of moisture being carried over with the steam leaving the steam generators. A tracer chemical, lithium hydroxide, is continuously injected into the main feedwater system until a 10-PPB concentration is achieved in the steam generator blowdown system. Main steam and steam generator blowdown samples are taken from existing sample points. Upon completion of the test, tracer injection is discontinued and the remaining lithium hydroxide will be removed using the steam generator blowdown system.

The percent moisture carryover is used in the reactor thermal power heat balance calculations. Presently, zero percent (0%) moisture carryover value is conservatively used, due to the lack of an accurate value. Determining the actual value will result in a more accurate reactor thermal power heat balance.
Summary of Evaluation:
The test will be conducted with Lithium levels equal to or greater than 5 ppb. It is anticipated by the amount of Lithium being injected that the concentration will be 10 ppb. A 14 ppb limit is established in the test procedure based on the action level 2 concentration of sodium (50 ppb).

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The test procedure reflects that if the concentration exceeds 14 ppb (equivalent to 50 ppb Sodium), the actions of procedure CH-SQ-13.018 will be followed for action level 2 and Lithium injection will be minimized in order to complete the test.
The procedure reflects the duration of the test to be 30 minutes to attain the greater than 5 ppb minimum Lithium concentration in the steam generators and then six samples taken at five minute intervals.
Lithium can be removed from the SGs through the Steam Generator Blowdown (SGB) System. It is anticipated the level one guideline will be exceeded for eight hours, which is within the guidelines for returning the Sodium concentration to below 5ppb within seven days.
Provisions are included in the IP2 Secondary Water Chemistry Program to ensure that the pH will be maintained at optimum levels or corrective actions are initiated to correct the condition. Procedure CH-SQ-13.018 provides the corrective action to adjust the pH through chemical addition. Cation conductivity is also listed as a steam generator control parameter that may be impacted. Action levels exist in the CH-SQ-13.018 to reduce the effects of elevated cation conductivity. The other generator control parameters in procedure CH-SQ-13.018 (chloride and sulfate) are not expected to be impacted by the test.

02-245-EV-00-RE	0	Use of Unit 3 Appendix R Diesel Generator to Satisfy Unit 2 Technical Specification 3.7.C.3

Activity Description:
The proposed change will establish the Unit 3 Appendix R Diesel Generator (ARDG) as "an alternate independent power system" (as specified in Unit 2 Technical Specification (TS) 3.7.C.3 if no Gas Turbines are OPERABLE. The ARDG currently satisfies Appendix R and SBO Safe Shutdown power requirements for Unit 3.

Summary of Evaluation:
No new equipment is being introduced to the site. The Safety Evaluation associated with Unit 3 MOD-85-03-004-ARDG documented that it was acceptable to power the 6.9 KV "Gas Turbine Bus" using the ARDG. The ARDG is periodically tested to ensure it can power the 6.9 KV "Gas Turbine Bus". The 6.9 KV "Gas Turbine Bus" is common between Unit 2 and Unit 3. Calculation FEX-00178-00 has documented that the ARDG can safely power the Unit 2 Alternate Safe Shutdown System (ASSS). Existing equipment (that is the EDGs) will continue to be operated as before. The response of equipment used to establish and maintain Safe Shutdown following a fire will be unchanged by using the ARDG and the three IP2 EDGs as a source of Alternate AC (AAC) power during a SBO event as shown in calculation FEX 00179-00 as "an alternate independent power system". The Design Basis for an AAC is

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satisfied without any GTs. Therefore, the proposed change is not expected to result in the introduction of any new accident or equipment malfunction. Nor is the consequence of any accident or malfunction of equipment expected to increase.

02-412-CL-00-RE	0	Containment Peak Pressure for a Postulated Steam Line Break

Activity Description:

UFSAR Section 14.2.5.6, "Containment Peak Pressure for a Postulated Steam Line Break", is based on analysis which credits the closure of the main feedwater (MFW) stop valve, BFD-5, to terminate the event. The peak containment pressure response has been re-analyzed without credit for the closure of the BFD-5 valve. The containment pressure response for the failure of the bypass feedwater control valve without credit for the closure of the bypass stop valve, BFD-90 has also been specifically analyzed. Further, the peak containment pressure resulting from the failure of the main feedwater control valve and with credit for the main feedwater stop valve (BFD-5) to close in 120 seconds has also been calculated. The UFSAR description and results are being revised to reflect the re-analysis (see UFSAR Section 14.2.5.6 and 14.3.5.1.1).

Summary of Evaluation:

The re-analysis of the main steam line break with failure of the MFW control valve and without credit for the MFW stop valve (BFD-5) results in a peak containment pressure of 46.7 psig. The analysis of the main steam line break with failure of the MFW bypass control valve and without credit for the MFW bypass stop valve (BFD-90) results in a peak containment pressure of 37.64 psig. The peak containment pressure resulting from the failure of the main feedwater control valve and with credit for the main feedwater stop valve (BFD-5) to close in 120 seconds was 36.8 psig. All these results remain below the containment design pressure limit of 47 psig.

02-420-CL-00-RE	0	Evaluation for a Potential Recritiality

Activity Description:

The licensing basis for IP2 will now credit the boron worth of inserted control rods and Xenon credit at the time of hot leg switchover to address the potential for recriticality due to sump dilution when realigning to hot leg recirculation. This action is in response

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to issues identified in Westinghouse Nuclear Safety Advisory Letter NSAL-94-016 Revision 2.

Summary of Evaluation:
The 50.59 Screen 'screened in' this activity since it is concluded that this activity effects post-LOCA subcriticality methodology described in the UFSAR. The 50.59 Evaluation reviewed the "The Method of Assuring Post-LOCA Subcriticality" and the "Method for Demonstrating Control Rod Insertion in Post-LOCA Evaluations" and concluded that these methodologies are either 'previously approved by the NRC' or 'not described in the UFSAR or its references'. Therefore, this activity is suitable for implementation.

02-498-EV-00-RE	0	Remote Examination and Removal of Foreign Objects from Steam Generator

Activity Description:
This evaluation allows operation of Indian Point 2 up to 5% of Reactor Power at a reduced Reactor Coolant System (RCS) pressure of 2000 psia (nominal). For purposes of this evaluation, the Reactor Power refers to the current rating of 3071.4 MWt or a 1.4% uprate power of 3115 MWt. This low RCS pressure operation is valid for Cycle 16, and for future cycles, the Safety Analysis Limits will be confirmed on a cycle specific basis as part of the Reload Safety Analysis Checklist process. The reduced pressure condition allows for reseating of Pressurizer Safety Valves in the event these valves should leak on a plant restart following an outage. This Safety Evaluation is to be implemented at the current power level only.

Summary of Evaluation:
The low RCS pressure at low power operation will not adversely affect the safe operation of the plant and does not require a change to the plant Technical Specifications. This low RCS pressure operation is valid for Cycle 16 at the current power level. This low RCS pressure operation is also valid for future cycles at the current power level by confirming the Safety Analysis Limits on a cycle specific basis as part of the Reload Safety Analysis Checklist process.